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Measurement of Dose-Equivalent Rates around a Cask and Monte Carlo Analysis with Actual Configuration of Fuel Basket

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Gamma ray and neutron dose-equivalent rate distributions are measured with an ionization-type gamma-ray survey meter and a moderator-type neutron survey meter respectively around a TN-12A spent fuel transport cask, and the measured dose-equivalent rate distributions are analyzed by the Monte Carlo method. Two models for the aluminum-alloy fuel basket are considered in the Monte Carlo code MCNP 4B. In one case the configuration of the basket with 12 spent fuel assemblies is modeled in detail and the other is the homogenized basket as employed in the Sn code DOT 3.5. In addition, the burn-up distribution is taken into account to generate source neutrons and gamma rays in the z-axis of the spent fuel assemblies in both cases. The essential difference in the dose-equivalent rates is obtained from the Monte Carlo calculations employing the homogenized model and the actual configuration of the basket. Due to employing the actual configuration, the gamma ray and neutron dose-equivalent rates reduce to 67 % and 80 %, respectively as compared with the homogenized model on the surface of the TN-12A cask. As the results, the good C/Es are obtained: for the neutron it is approximately 1.0 and 1.25 for the gamma ray it is at the center of the cask surface, respectively. The effect of the burn-up distribution appears clearly at the off-center cask surface, and in particular, the neutron dose-equivalent rates come close to the measured ones.

KEYWORDS: Spent fuel, transport cask, Monte Carlo analysis, MCNP 4B code, gamma ray, neutron, dose-equivalent rate, survey meter, fuel basket, fuel assembly, homogenized model, actual configuration, burn-up distribution

I. Introduction

Accurately estimating the distribution of radiation dose-equivalent rates around a spent fuel-shipping cask is fundamental to the analysis for the safety of the cask and the reliability of its shielding design. When measured the dose-equivalent rates exceed the calculated values, a cask cannot contain its expected amount of spent fuels. On the other hand, if the measured dose-equivalent rates are below the estimations, the efficiency of a cask is low due to over-design. The validity of the Monte Carlo method has been demonstrated in shielding analysis of a spent-fuel transport[1,2] cask and a shipping vessel[3]. However, the Monte Carlo has not been yet employed as a means of the cask shielding design.

In spite of a restriction for geometrical expression, the two-dimensional discrete-ordinates Sn code DOT 3.5[4] has been employed as a typical code for shielding analysis of a cask. In the cask shielding analysis, the restriction is indicated noticeably in modeling of an aluminum-alloy fuel basket. For example, 12 PWR (Pressurized Water Reactor) spent fuel assemblies are loaded in the fuel basket of TN-12A spent fuel transport cask[3]. However, gamma rays and neutrons are generated uniformly in the homogenized region of the spent fuels and the basket by the calculation. In the current study, the Monte Carlo method is employed for the shielding analysis of the TN-12A cask, and the actual geometry of the basket with spent fuels, which is originally used in the criticality analysis, is employed in the shielding analysis. In addition, the burn-up distribution is taken into account to generate source neutrons and gamma rays in the z-axis of the spent fuel assemblies. The profile of the relative burn-up distribution is like a trapezoid and both the top and the end are less than the middle in the z-axis. The burn-up effect is considerable especially for the neutron-source generation in the spent fuels and leads to decreasing the neutrons around the top and the bottom trunnions of the cask.

Gamma ray and neutron dose-equivalent rate distributions are measured with an ionization-type survey meter and a moderator-type survey meter respectively around the TN-12A spent fuel transport cask before loaded in a shipping vessel, and the dose-equivalent rate distributions are analyzed by the Monte Carlo method in detail. The essential difference in the calculated dose-equivalent rates are obtained by the calculations by using the continuous energy Monte Carlo code MCNP 4B[5] employing the homogenized model and the actual configuration of the aluminum-alloy fuel basket with spent fuels. The burn-up distributions of the spent fuel assemblies are taken into account in these calculations.

II. Modeling of the Cask

Even now, the homogenized model of the aluminum-alloy fuel basket has been employed in the shielding calculation for the safety analysis report of the cask. The reason is that the two-dimensional discrete ordinates Sn code DOT 3.5 has been employed as the main code for the shielding calculation, even though the Monte Carlo code has been used in the criticality analysis in which the actual configuration of the basket is

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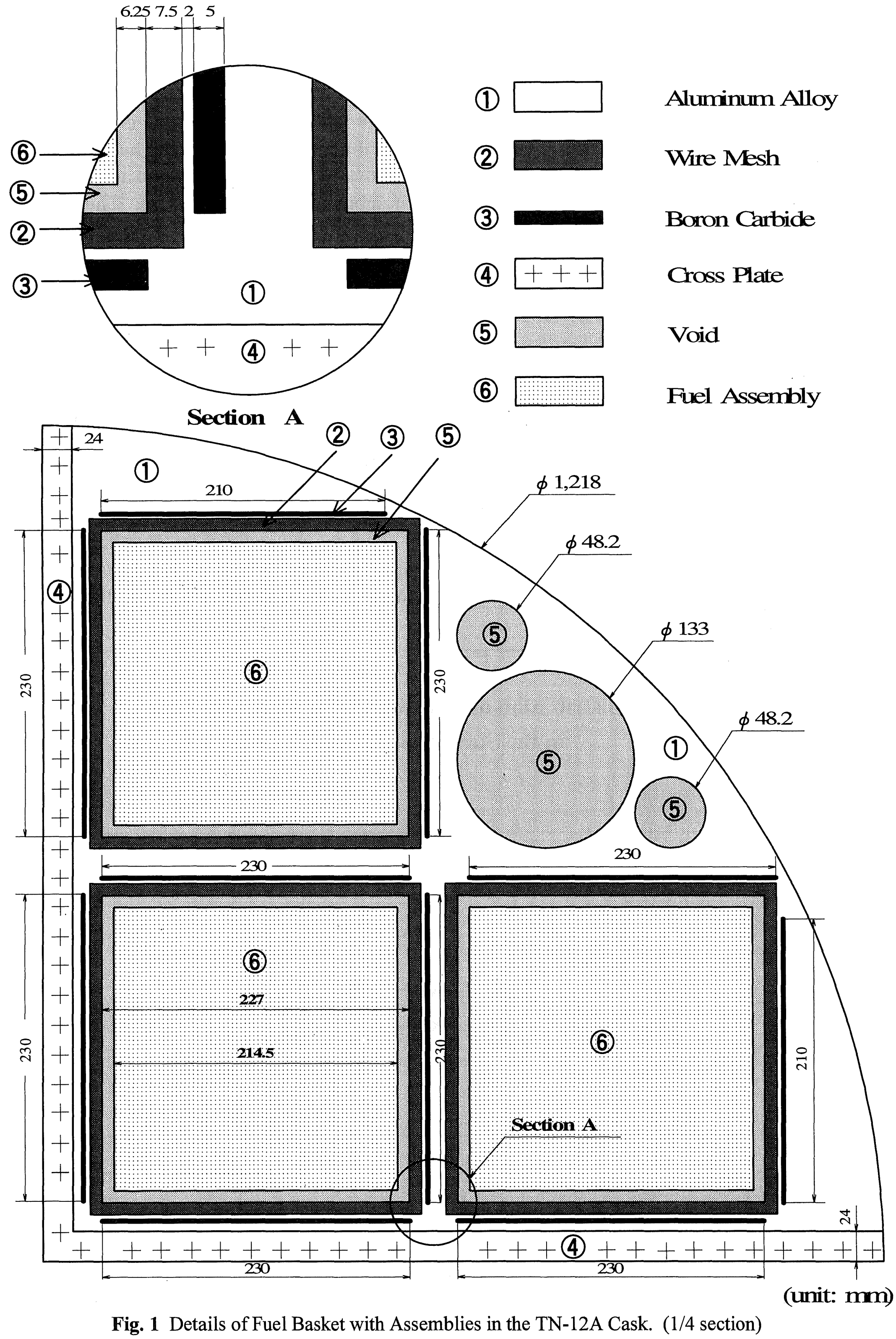

Fig. 1 Details of Fuel Basket with Assemblies in the TN-12A Cask. (1/4 section)

employed. Accordingly, it is not so difficult to use the actual geometry with the Monte Carlo for the shielding analysis. When taking into account the geometry, gamma ray and neutron sources are generated in the spent fuel assemblies, and as shown in **Fig. 1**, the basket and the other constructions around the assemblies have the effects of shields. The generating locations of source gamma rays and neutrons are limited in the assemblies, and the source generating locations are to be more inner part than that of the homogenized model. Therefore, the Monte Carlo calculations with the actual configuration as shown in Fig. 1 are expected that the dose-equivalent rates are lower than that with the homogenized model and that more accurately estimating will be achieved.

Figure 1 is 1/4 section of the basket loaded in the TN-12A cask. The aluminum-alloy basket can contain the 12 PWR fuel assemblies. The wire mesh and the boron carbide plate are surrounding the assemblies. The former is as a shock absorber and the later is for criticality safety.

Table 1 Specification of Spent Fuel Assemblies Loaded in the TN-2A Cask

Reactor Type	PWR
Burn-up	
Middle	30,500 MWd/MTU*
Top and End	26,000 MWd/MTU
Specific Power(middle)	27 MW/MTU
Cooling Time	660 days
Enrichment	3.1 wt%
Neutron Source (k_{eff} = 0.3)	
Middle	1.90×10^8 n/s/assembly
Top and End	4.75×10^7 n/s/assembly
Gamma-ray Source	
Middle	5.69×10^{15} p/s/assembly
Top and End	2.42×10^{15} p/s/assembly

In addition, the burn-up distribution is taken into account to generate source neutrons and gamma rays in the z-axis of the spent fuel assemblies. The data on the relative burn-up distribution is taken from the burn-up of the PWR fuel. The relative burn-up at both the top and the end of a fuel assembly are about 85 % of the middle. Then the gamma-ray source intensity at both the ends is reduced to 0.85, but the neutron source intensity depends on the burn-up more than that of the gamma ray and it reduces to 1/2 of the middle. The specification of the spent fuel assemblies loaded in the TN-12A cask is indicated in **Table 1**. Taking into account the burn-up distribution of the spent fuels, the neutron dose-equivalents decrease considerably at the off center of the cask, especially around the top and the bottom trunnions.

III. Monte Carlo Analysis and Comparison with Experiment

Gamma ray and neuron dose-equivalent rate distributions around the TN-12A cask were measured with the Oyogiken ionization-type survey meter and the Studvik moderator-type

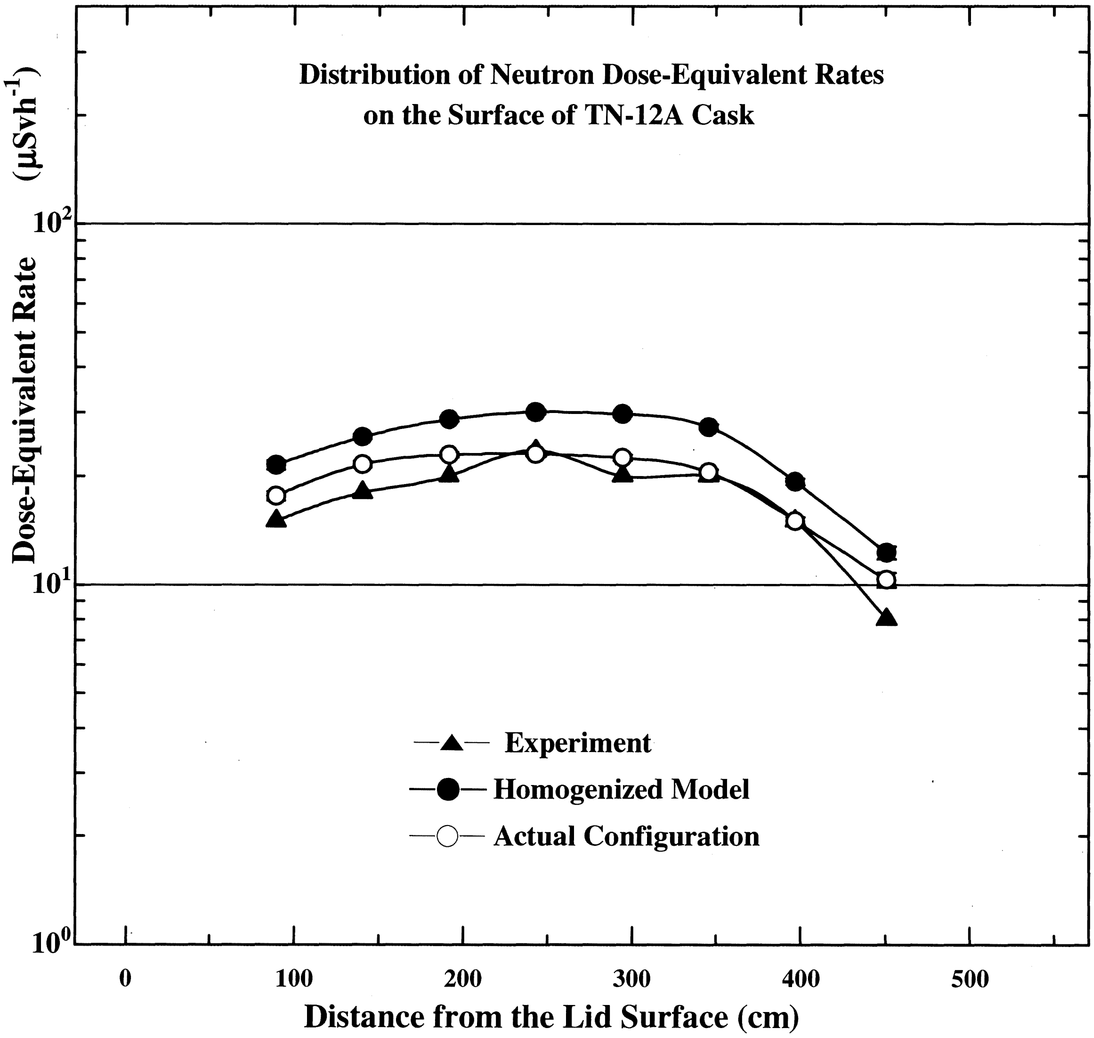

Fig. 2 Comparison of Dose-Equivalent Rate Distribution between Measured and Monte Carlo Calculations on the Surface of TN-12A Cask

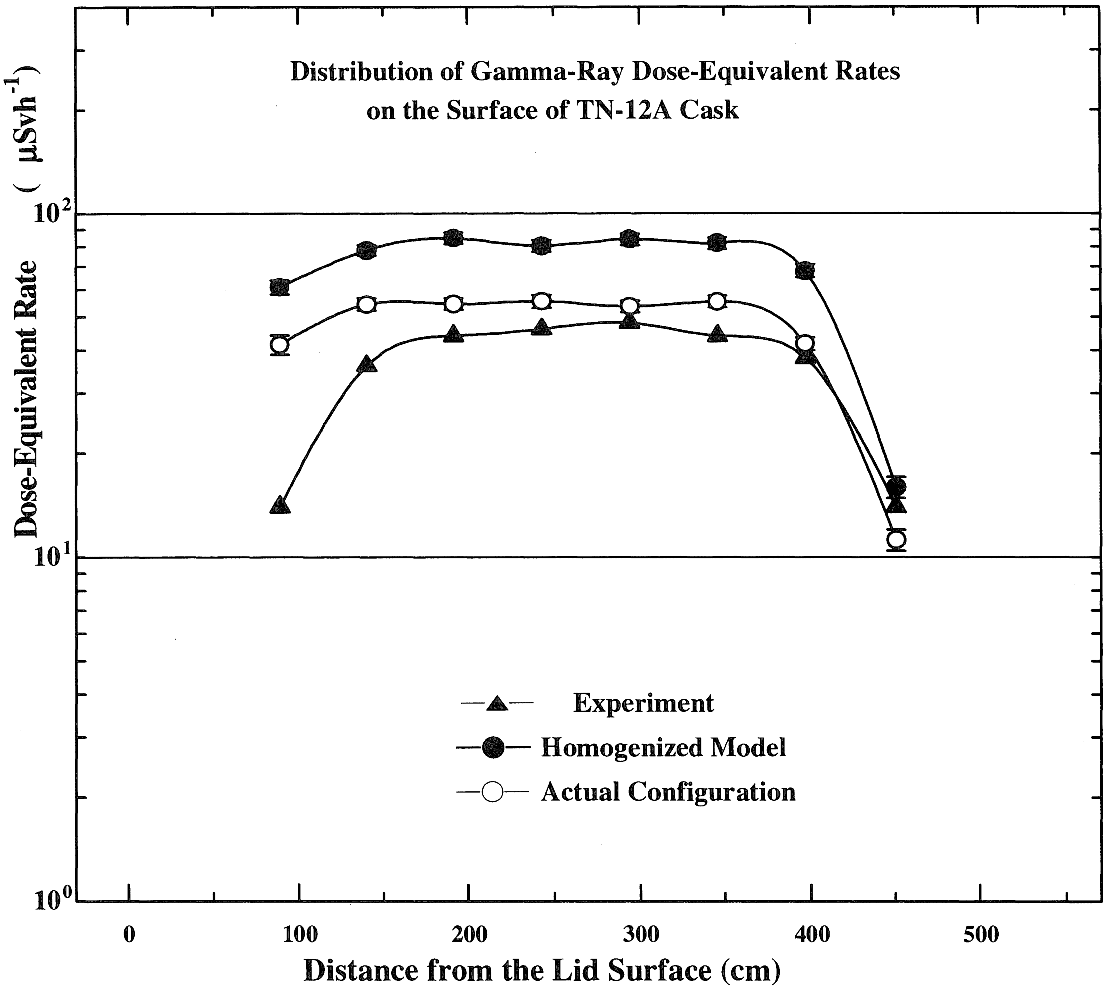

Fig. 3 Comparison of Dose-Equivalent Rate Distribution between Measured and Monte Carlo Calculations on the Surface of TN-12A Cask

survey meter, respectively before loaded in a vessel. The maximum dose-equivalent rate appeared at the center of the cask surface and it was 23.5 μSv/h for the neutron and 48 μSv/h for the gamma ray.

The maximum burn-up of the fuels installed in the TN-12A cask was 30,500 MWD/MTU in the middle of the fuel assemblies and the burn-up was reduced to 26,000 MWD/MTU at the top and the end of the assemblies when considering the burn-up distribution[6]. The effective length of the fuel assembly is 385 cm. The burn-up of 30,500 MWD/MTU in the middle corresponds to 308 cm (8/10 of the effective length) and that of 26,000 MWD/MTU at the top and the end of the assembly corresponds to 38.5 cm (1/10 of it). The burn-up distributions are taken into account in both calculations of the actual configuration and the homogenized model. The comparison of neutron and gamma-ray dose-equivalent rate distributions on the TN-12A cask surface between the measured and the Monte Carlo calculations are shown in **Fig. 2** and **Fig. 3**, respectively.

The gamma ray and neutron dose-equivalent rates with the actual configuration reduce to 67 % and 80 %, respectively as compared with the homogenized model at the center of the TN-12A cask surface. As the results, the good C/Es are obtained: for the neutron it is approximately 1.0 and 1.25 for the gamma ray at the center of the cask surface in Fig. 2 and Fig. 3, respectively. Particularly, the calculated neutron dose-equivalent rates decrease like a cosine curve at the off-center part and come close to the measured dose-equivalent rate distribution through the cask surface. The gamma-ray distribution with the actual configuration indicates also good agreement with the experiment except at the top trunnion: at 100 cm in Fig. 3. The cask structures around the trunnion are so complicated that modeling of the structures might not be enough. The structures are made of carbon steel or stainless steel which materials are effective shields for gamma rays. As a result, it leads to over estimation of the Monte Carlo calculations as compared with the measurements around the top trunnion.

IV. Concluding Remarks

The following concluding remarks are obtained from the measurements of dose-equivalent rates around the cask and the intensive Monte Carlo calculations taking into account the actual configuration of the basket and the burn-up distribution of the spent fuels.

1. The Monte Carlo calculations of the gamma ray and neutron

dose-equivalent rates with the actual configuration of the fuel basket reduce to 67 % and 80 %, respectively as compared with the homogenized model of it on the surface of the TN-12A cask. As the results, the good C/E is obtained: for the neutron it is approximately 1.0 and for the gamma ray it is 1.25 at the center of the cask surface.

2. Particularly, the calculated neutron dose-equivalent rates decrease like a cosine curve at the off-center part and come close to the measured dose-equivalent rate distribution through the cask surface.

3. Through the Monte Carlo calculations it is established that the employment of the actual configuration of the aluminum-alloy fuel basket and the burn-up distribution of the spent fuels are quite essential to obtain the radiation dose-equivalent rates of the cask accurately. When the detailed configuration of the basket and the burn-up distribution are informed, it is recommended to take those data in the Monte Carlo calculations.

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